NON-PUBLIC?: N
ACCESSION #: 8908310024
LICENSEE EVENT REPORT (LER)

FACILITY NAME: SAN ONOFRE NUCLEAR GENERATING STATION UNIT 1

PAGE: 1 OF 7

DOCKET NUMBER: 05000206

TITLE: MANUAL REACTOR TRIP FOLLOWING A LOSS OF FEEDWATER TO ONE STEAM
GENERATOR AS A RESULT OF MIS-COMMUNICATION
EVENT DATE: 07/24/89 LER #: 89-019-00 REPORT DATE: 08/23/89

OPERATING MODE: 1 POWER LEVEL: (10) 076

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION
50.73(a)(2)(ii)

LICENSEE CONTACT FOR THIS LER:
NAME: H. E. Morgan, Station Manager TELEPHONE: 714 368-6241

COMPONENT FAILURE DESCRIPTION:
CAUSE: SYSTEM: COMPONENT: MANUFACTURER:
REPORTABLE NPRDS:

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

On July 24, 1989, at 1216 while at 76% power, Unit 1 was manually tripped due to a loss of feedwater flow to Steam Generator (SG) "A" and resultant low level. The loss of feedwater flow occurred during performance of a test of the SG "A" high level alarm. At 1215, operators authorized maintenance technicians to perform testing of the SG "A" high level alarm and Steam Flow/Feed Flow Mismatch Reactor Trip alarm. During the high level alarm portion of the test, the SG "A" high level alarm annunciated as anticipated. The high level alarm was promptly followed by the SG "A" Steam/Feed Mismatch alarm, which was not anticipated at this point in the test sequence. After observing that SG "A" levels were rapidly decreasing and the SG "A" main feedwater Flow Control Valve (FCV) had tripped closed, operators unsuccessfully attempted to open the FCV and, in accordance with procedures, manually tripped the reactor. SG "A" may have dried out for a brief period shortly after the reactor had been tripped until the auxiliary feedwater

system actuated at 1217. All required systems functioned normally.

The effects of a recent design change on the FCV circuitry were not recognized to result in a loss of feedwater flow and were, therefore, not transferred into station procedures or operator training. The root cause is attributed to weaknesses in SCE's processes for ensuring that design change information is adequately incorporated into procedures.

The methods by which design change information is prepared and reviewed for incorporation into Station Procedures will be reviewed by a multi-disciplinary group to determine appropriate enhancements and corrective actions necessary to assure recognition of impacts. The SG high level alarm test will be performed only while the unit is shutdown. Other recent design changes were reviewed for similar discrepancies with none being found.

END OF ABSTRACT

Plant: San Onofre Nuclear Generating Station
Unit: One
Reactor Vendor: Westinghouse
Event Date: 07-24-89
Time: 1216

A. CONDITIONS AT TIME OF THE EVENT:

Mode: 1, Power Operation
RCS Temperature: 549 degrees F

B. BACKGROUND INFORMATION:

1. System Information

There are three Narrow Range (NR) Steam Generator (SG) SG! Level Instrumentation LT! channels and two Wide Range (WR) channels (one WR channel is hot calibrated and the other is cold calibrated) for each SG. One of the NR channels is utilized to provide, in part, a high level SG alarm and to automatically trip closed the associated feedwater flow control JB! valve (FCV!). The purpose of closing the FCV on a high level is to preclude over-filling the SG.

Unit 1 is provided with two Safeguards Load Sequencing Systems (SLSS) JE! which actuate required equipment in the sequence necessary to protect the plant in the event that safety injection or containment isolation is required.

2. Design Change

Prior to the Cycle 10 refueling outage, the logic which protected the SGs from potential overfilling required actuation of both a SLSS on a safety injection actuation signal and a high SG level alarm bistable. With this design configuration, a SG high level alarm would not, by itself, trip the associated SG FCV closed or block manual control of the valve.

During the Cycle 10 refueling outage, the automatic trip of a SG FCV was changed so that the FCV tripped closed on either: (1) a SG high level provided by the SG's high level alarm bistable, or (2) a SLSS actuation on safety injection. This design change also prevented reset and control of the SG FCVs until the SG high level condition had been corrected. The purpose of changing the FCV trip circuitry is to enhance SG overfilling protection.

3. Design Change Process

Administrative procedures describing the development, review, approval, and issuance of Proposed Facility Changes (PFCs) and Design Change Packages (DCPs) establish the controls necessary to verify compliance with the plant design bases. When it is identified that a

plant modification is required or desired, a DCP is developed which contains the actual document and drawing changes and is normally routed for review with the PFC (although it receives different reviews than the PFC). A PFC is also developed which includes: 1) a description of the change, including the technical basis for the change; 2) a list of the design and operating documents which are affected; and 3) evaluations which describe the impact of the change on design parameters, operations, and plant safety. Since the PFC contains the general explanation of the design change, it becomes the controlling document for the change. The DCP is normally attached to, or referenced, by the PFC. DCPs receive: 1) an independent engineering review, 2) an interdisciplinary engineering review, 3) supervisory engineering reviews, and 4) a Quality Assurance (QA) review. PFCs receive: 1) supervisory engineering reviews, 2) a station interdepartmental review, 3) safety review, and 4) a QA review.

The safety and QA reviews are normally completed after implementation of the PFC/DCP.

C. DESCRIPTION OF THE EVENT:

1. Event:

On July 24, 1989, at 1216 while at 76% power, Unit 1 was manually tripped due to a loss of feedwater flow to Steam Generator (SG) "A" and resultant low level. The loss of feedwater flow occurred during the performance of a test of the SG "A" high level alarm. At 1215, operators authorized maintenance technicians to perform testing of the SG "A" high level alarm and Steam Flow/Feed Flow Mismatch Reactor Trip alarm. During the high level alarm portion of the test, the SG "A" high level alarm annunciated as anticipated. The high level alarm was promptly followed by the SG "A" Steam/Feed Mismatch alarm which was not anticipated since the maintenance technicians had not yet started the Steam/Feed Flow Mismatch alarm portion of the test. The operators observed that a light, which indicates that SG "A" feedwater flow control valve (FCV-456) had tripped closed, was illuminated and that the remaining NR channels and the WR level indications were rapidly decreasing 1_/. After an unsuccessful attempt to reset and manually open FCV-456 from the control room in accordance with procedures, the reactor was tripped as described above.

Based on the post-trip evaluation of WR level computer data, it is be
ieved that SG "A" may have dried out a few seconds after the reactor had been tripped. At 1217, the Auxiliary Feedwater System (AFW) BA! actuated in accordance with the design. The AFW actuation was initiated by low SG level in two-of-three SGs due to the loss of feedwater to SG "A" and level "shrink" which occurs following a

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1_/ A low level alarm was not received since both the SG high and low level annunciator alarms operate from the same narrow range channel which was out of service for the test.

reactor trip. All AFW actuation channels actuated in accordance with the design.

2. Inoperable Structures, Systems or Components that Contributed

to the Event:

None.

3. Sequence of Events on July 24, 1989:

TIME ACTION

-1215 Operators approve performance of the SG "A" high level alarm and steam/feed mismatch alarm test.

Operators receive SG "A" high level alarm as anticipated.

1215 WR SG level post-trip printout first indicates a decreasing SG "A" level.

between
1215
and Operators:
1216
(1) Receive a SG "A" Steam/Feed Flow Mismatch Alarm, which was not anticipated during this part of the test.

(2) Observe a rapidly decreasing SG "A" level on the NR and WR level instrumentation.

(3) Unsuccessfully attempt to reset and open FCV-456.

(4) Trip the reactor on SG "A" low level.

1217 Auxiliary Feedwater flow to SG "A" is first indicated on post-trip printout.

4. Method of Discovery:

As described in the above sequence of events.

5. Personnel Actions and Analysis of Actions:

Operations personnel response was appropriate, timely (78 seconds from the first recorded drop in WR SG level until the first recorded reactor power decrease due to the trip), and in full conformance with all applicable procedures. Based upon a review of this event, it became apparent that procedural guidance to the operators for loss of feedwater flow to a SG could be enhanced so that the reactor would be

tripped sooner. As described below in Part E.2.a, such enhancements are being developed.

6. Safety System Responses:

The AFW and the reactor protection system JC) performed as required by the design.

D. CAUSE OF THE EVENT:

1. Immediate Cause:

Performance of the SG "A" high level alarm channel test tripped FCV- 456 closed. This was the first time the test had been performed at power after the SG FCV circuitry design was changed to provide for FCV closure on either SG high level or the SLSS.

2. Intermediate Cause:

The effects of the design change on the high level alarm test were not recognized to result in a loss of feedwater flow and were, therefore, not transferred into station procedures or operator training. Consequently: (1) the issue of how the high SG level alarm test could be performed with the unit at power was not addressed nor was the test procedure revised; and (2) the ability to control an affected SG FCV was not addressed in operator training or in the SG low level operating instruction.

3. Root Cause:

The cause of this failure was a mis-communication of design change information from the design organization to the station organizations who must recognize and use the change information. The mis-communication resulted from a mismatch in the expectations of the two types of organizations about the type and identification of design change information which may impact the station organizations. Additionally, there is no training or formal guidance which would enhance the ability of non-engineering PFC reviewers to identify design change impacts within their area of responsibility.

E. CORRECTIVE ACTIONS:

1. Corrective Actions Taken:

a. The SG high level alarm test interval has been changed from once per 31-days to once per refueling. This frequency has been reviewed and determined to be acceptable.

b. All design changes implementing Cycle 10 changes to SG level indication and control, other control functions based on SG parameters, or enhancements to these control and instrumentation

functions to meet single failure criteria, have been reviewed to ensure that appropriate procedure changes have been implemented. No discrepancies were noted.

2. Planned Corrective Actions:

a. The SG low level abnormal operating instruction will be modified to provide more explicit direction to operators regarding actions to be taken in response to a loss of feedwater resulting from a SG feedwater FCV failure. The changes to this instruction should result in the reactor being tripped with ample SG water inventory.

b. During the review of the causes of this event, it was recognized that the previously completed corrective action discussed in Part G.2 could not, in itself, prevent recurrence of a failure to recognize design change impacts.

As a result, SCE will initiate a review of the process of communicating design change information from the design organization to station organizations. The specific objective of this review is to identify specific changes to design and reviewing organization procedures, resources and practices, and to identify any required training which may be necessary to communicate the impact of design changes. The review will be performed by representatives of the design organization and those site organizations who must identify design change impacts within their area of responsibility.

F. SAFETY SIGNIFICANCE OF THE EVENT:

This event has no safety significance since all components operated in conformance with the design and the unit remained within the bounds of all applicable analyses.

G. ADDITIONAL INFORMATION:

1. Component Failure Information:

Not applicable.

2. Previous LERs for Similar Events:

a. LER 88-006 (50-206) reported, in part, design implementation deficiencies in the backup nitrogen system as the result of inadequate implementation of design requirements in operating and maintenance procedures.

b. LER 88-010 (50-361) reported a condition in which both emergency chillers were rendered inoperable as a result of not addressing freon level as a critical design parameter.

Corrective actions have been implemented which address the issues identified in the above LERs. The two completed corrective actions applicable to the cause of this event were: (1) enhancement of the design control procedure to facilitate recognition of design change impacts on operations and maintenance activities; and (2) restructuring of the nuclear design organization so that a single system engineer is cognizant, and is the focal point, for all aspects of the assigned system. The system engineer is now available, as a single point of contact, to answer questions about the impacts of design changes.

However, since these corrective actions were implemented after completion of the design change which resulted in this event, they could not have prevented the event.

3. Results of NPRDS Search:

Not applicable.

4. Other additional information concerning the possible SC "A" dryout:

Dryout of the Unit 1 SGs has been previously analyzed by Westinghouse and was transmitted to the NRC by a letter from K. P. Baskin (SCE) to D. M. Crutchfield (NRC), dated June 3, 1982. The Westinghouse analysis (which addressed a single SG dryout event) determined that the potential thermal shock to the SG vessel and tubes was acceptable for far more severe conditions (i.e., higher flowrate and colder water) than was experienced during this event. Additional analyses are being performed to determine the actual number of dryout events that could occur without having an adverse impact on the SGs.

The leak rate from the reactor coolant system to the secondary system has not increased as a result of this event, confirming that SG primary to secondary leakage integrity was not affected by the event.

Evaluation of the effects of this transient on reactor power distribution has determined that the power distribution remained well within analyzed limits during this event.

Inspection and evaluation of data from previously installed linear displacement transmitters indicates that the SG "A" main feedwater line was not damaged as the result of the transient.

As discussed above in Part E.2.a, the SG low level operating instruction will be revised to provide explicit direction to operators in response to a loss of feedwater resulting from a SG feedwater FCV closure. This guidance will provide additional protection against SG dryout.

ATTACHMENT 1 TO 8908310024 PAGE 1 OF 1

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AUGUST 23, 1989

U. S. Nuclear Regulatory Commission

Document Control Desk
Washington, D.C. 20555

Subject: Docket No. 50-206
30-Day Report
Licensee Event Report No. 89-019
San Onofre Nuclear Generating Station, Unit 1

Pursuant to 10 CFR 50.73(d), this submittal provides the required 30-day written Licensee Event Report (LER) for an occurrence involving a manual reactor trip. Neither the health and safety of plant personnel or the public was affected by this occurrence.

If you require any additional information, please so advise.

Sincerely,

Enclosure: LER No. 89-019

cc: C. W. Caldwell (USNRC Senior Resident Inspector, Units 1, 2 and 3)
J. B. Martin (Regional Administrator, USNRC Region V)
Institute of Nuclear Power Operations (INPO)

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